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Fast Reactor Physics and Core Design

NRC Topical Seminar on Sodium Fast Reactors
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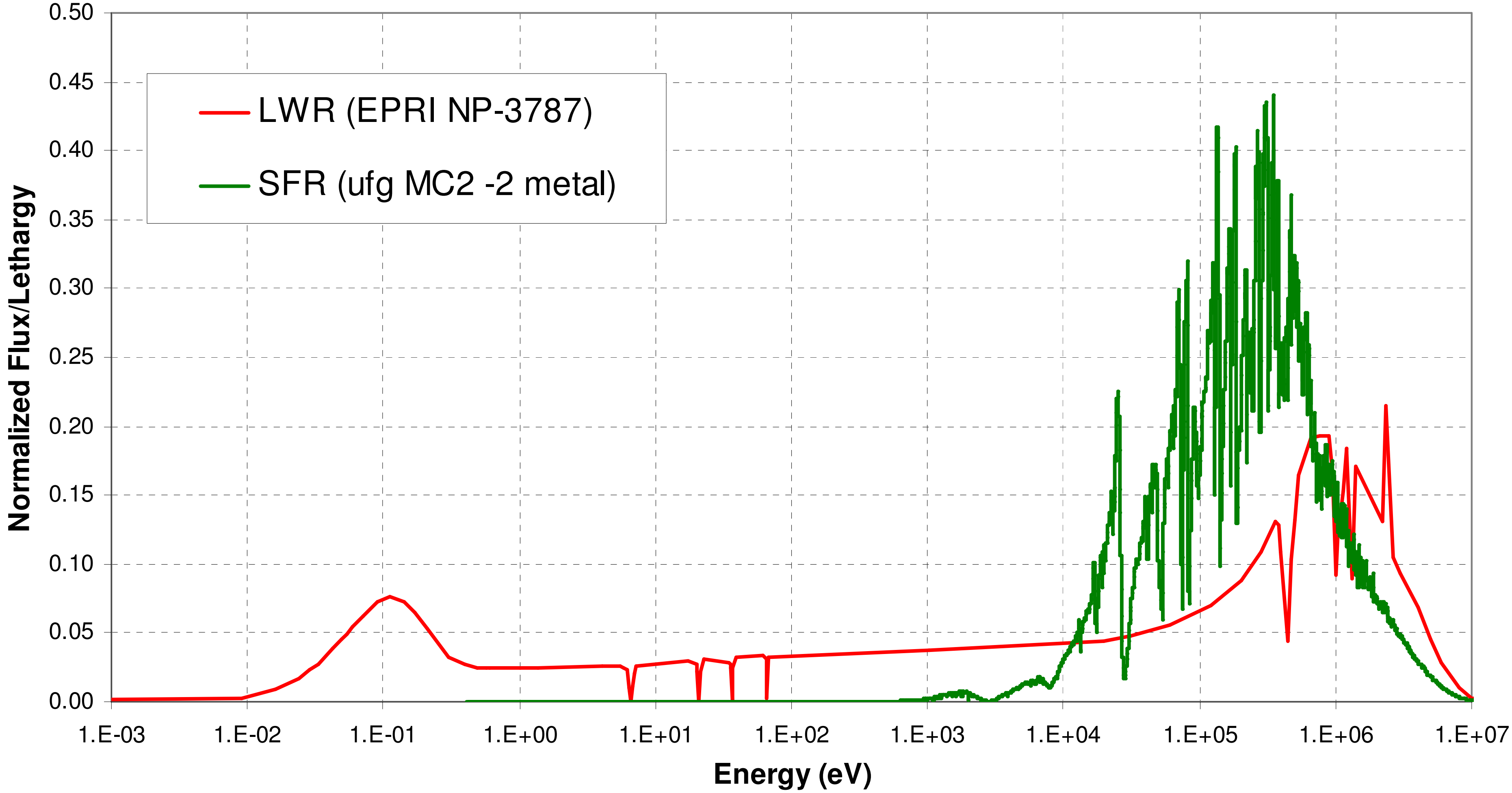
Outline

- Fast Reactor Physics
  - Contrast LWR physics and different fast reactor types
  - Important phenomena and modeling/data challenges
  - Impact on fuel cycle performance
  - *Brief* overview of existing methods and codes
- Sodium-cooled Fast Reactor (SFR) Core Design
  - Contrast to LWR design parameters
  - Typical reactor configuration
  - Typical reactor performance
- SFR Reactivity Coefficients
  - Identification of physics for each feedback
  - *Brief* discussion of safety implications

Comparison of LWR and SFR Spectra



- In LWR, most fissions occur in the 0.1 eV thermal "peak"
- In SFR, moderation is avoided – no thermal neutrons

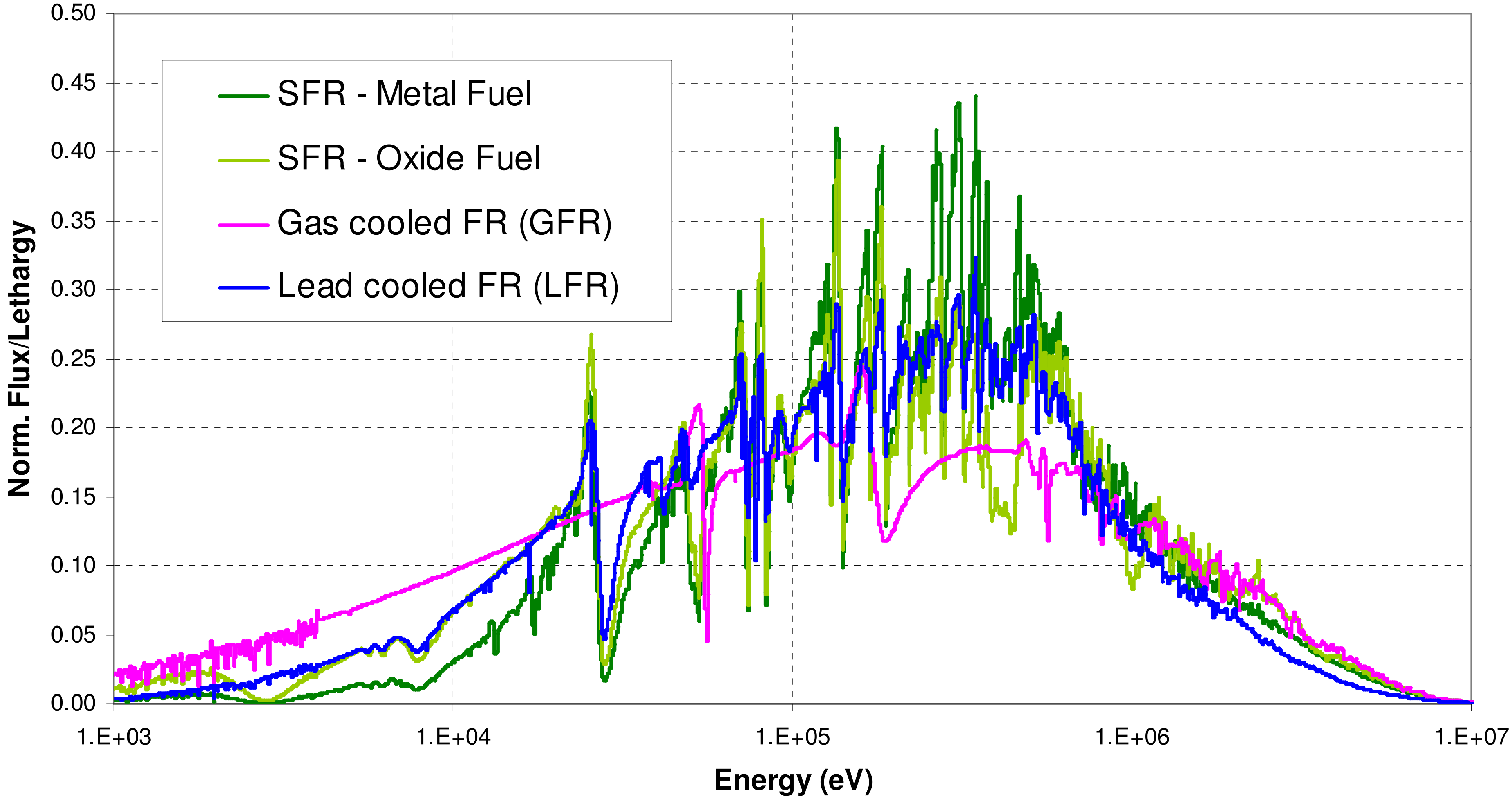
Neutron Moderation Comparison

- Significant elastic scattering of the neutrons in both spectra
- In FRs, neutron moderation is avoided by using high A materials
  - Sodium is most moderating
- In LWRs, neutrons are moderated primarily by hydrogen
- Oxygen in water and fuel also slows down the neutrons

	σ_s (barn)	N (#/barn·cm)	$\xi\Sigma_s$ (cm^{-1})
TRU	4.0	3.2E-03	1.1E-04
U	5.6	5.6E-03	2.7E-04
Zr	8.1	2.6E-03	4.6E-04
Fe	3.4	1.9E-02	2.3E-03
Na	3.8	8.2E-03	2.7E-03
H	11.9	2.9E-02	3.5E-01

- Slowing-down power in FR is ~1% that observed for typical LWR
- Thus, neutrons are either absorbed or leak from the reactor before they can reach thermal energies

Comparison of Fast Reactor Spectra



- Also, spectral differences between fast reactor concepts
- At high energy (>1 MeV) lead is effective inelastic scattering material
- Low energy tail caused by moderating materials (next viewgraph)

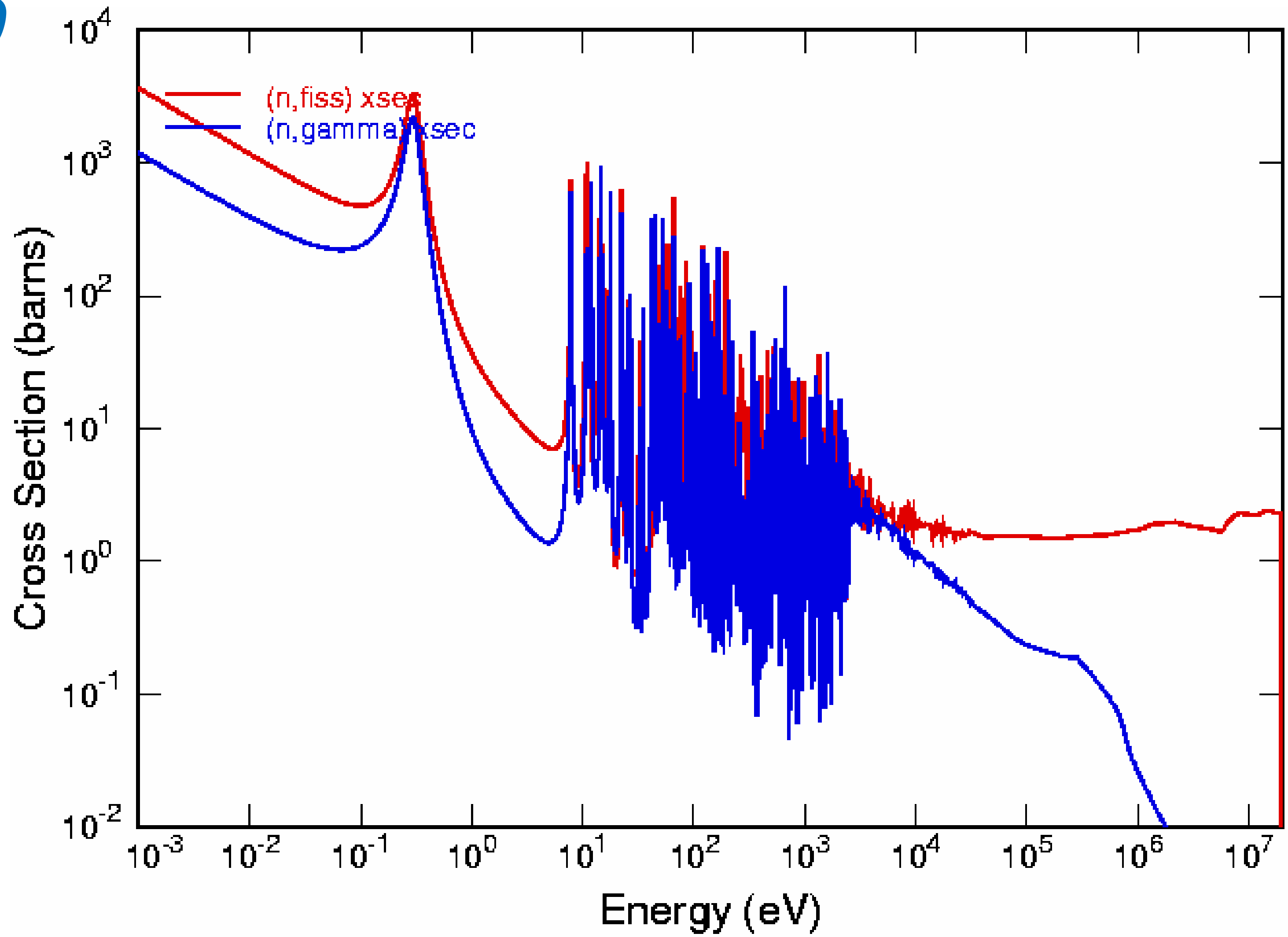
Fast Reactor Moderating Materials

- Lead has highest scattering of the neutrons, but little moderation
- Oxide fuel in SFR leads to a significant moderation effect – resonances also observed
- LFR designs also utilize nitride or oxide fuel forms
- In modern GFR designs, SiC matrix for fuel is utilized, with most moderation of the FR cases

	σ_s (barn)	N (#/barn·cm)	$\xi\Sigma_s$ (cm^{-1})
TRU	4.0	3.2E-03	1.1E-04
U	5.6	5.6E-03	2.7E-04
Zr	8.1	2.6E-03	4.6E-04
Fe	3.4	1.9E-02	2.3E-03
Na	3.8	8.2E-03	2.7E-03
O	3.6	1.4E-02	5.8E-03
Pb	8.6	1.6E-03	1.3E-04
C	3.9	1.6E-02	1.0E-02
He	1.7	3.1E-04	2.3E-04

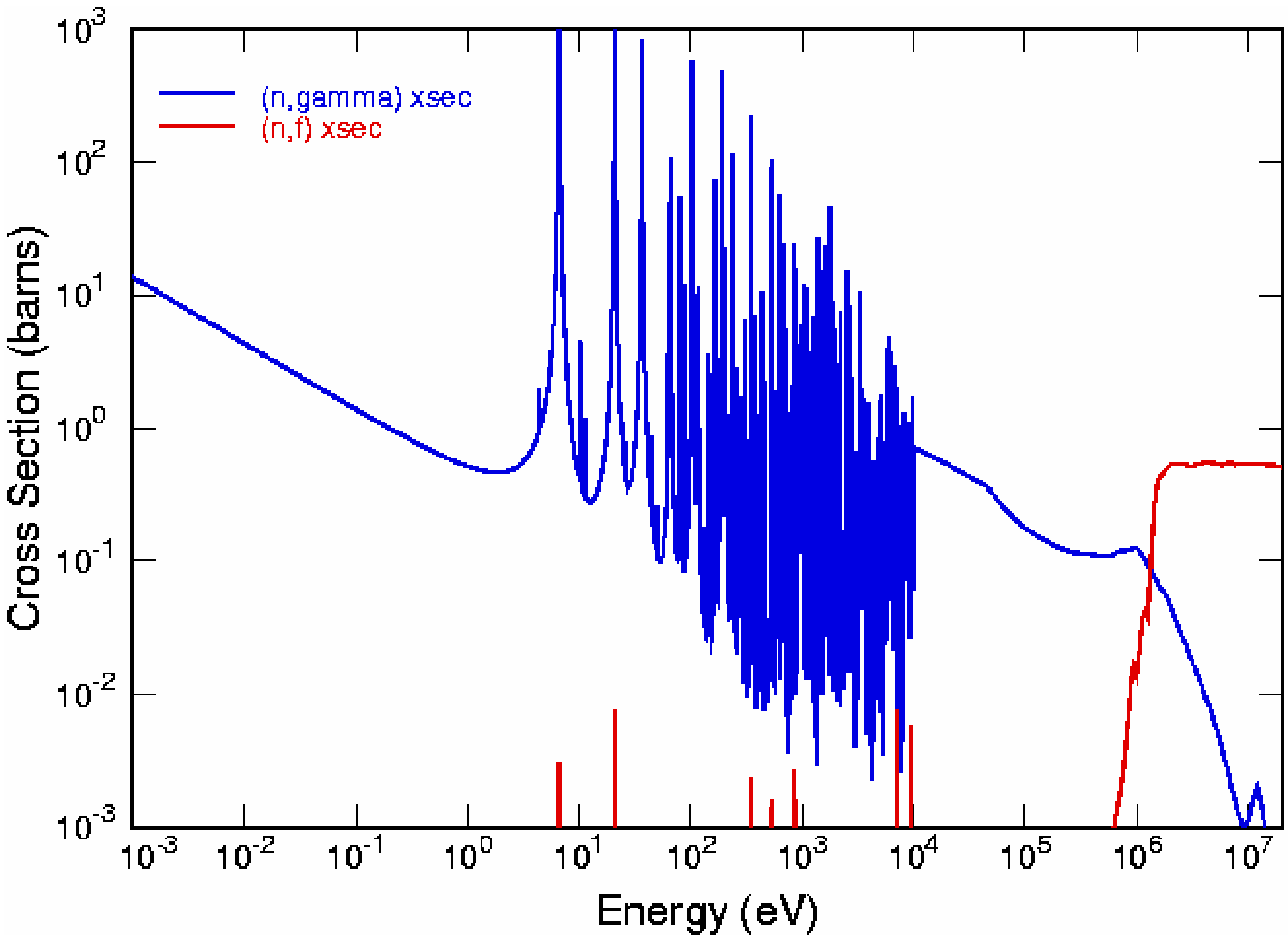
- Net result is that the SFR-metal has the hardest neutron spectrum
  - SFR-oxide and LFR similar with slightly more moderation
- GFR (with silicon-carbide matrix) has significant low energy tail

Spectral Variation of Neutron Cross Sections Pu-239



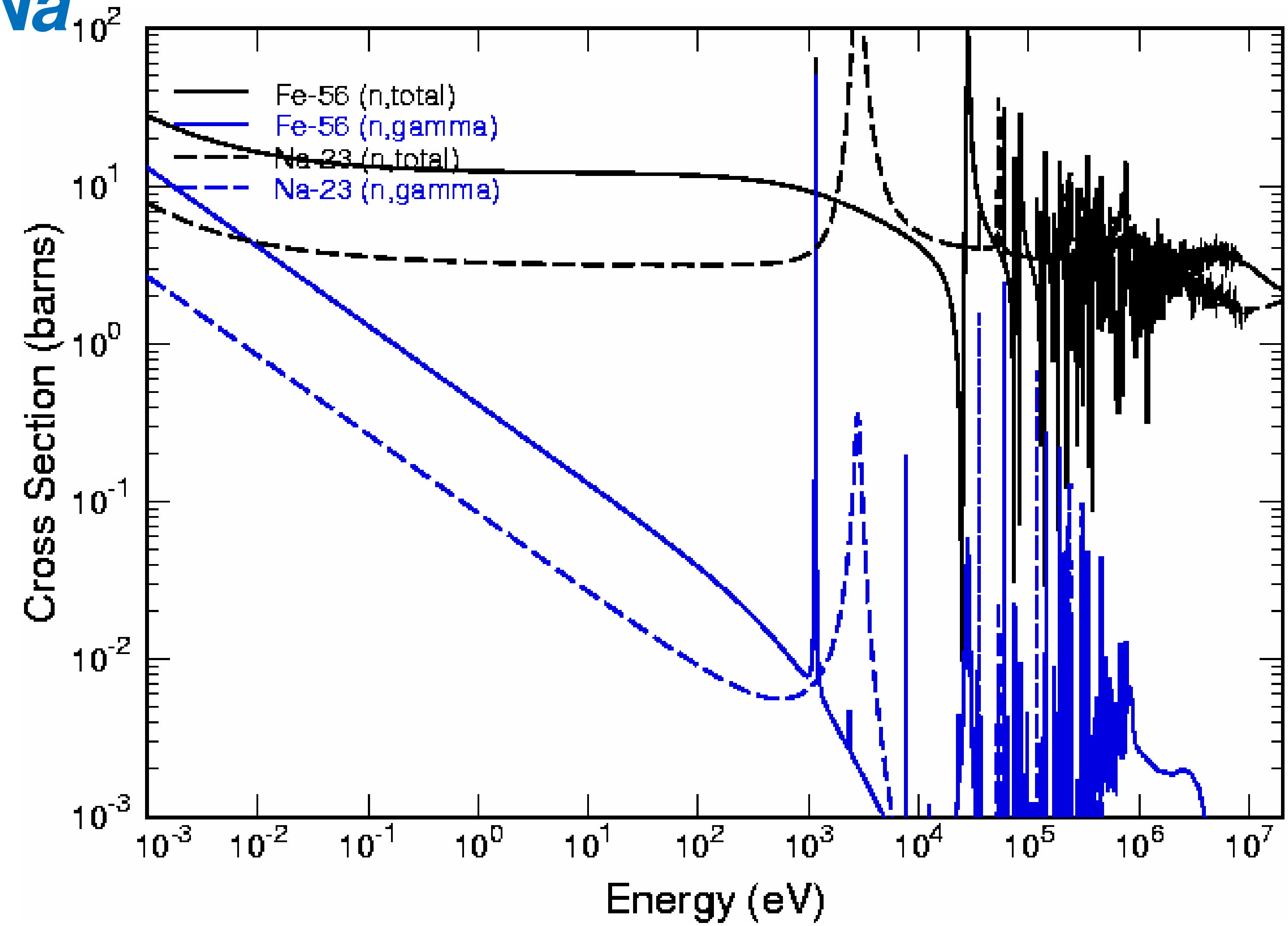
- Fission and capture cross section >100X higher in thermal range
- Sharp decrease in capture cross section at high energy

Spectral Variation of Neutron Cross Sections U-238



- Much smaller thermal increase in capture (~10X)
- Unresolved resonance range begins at ~10 keV
- Threshold fission at ~1 MeV

Spectral Variation of Neutron Cross Sections Fe and Na



- Capture cross sections much higher in thermal range
- Significant scattering resonance structure throughout fast range

Implications of Fast Spectrum Physics

- Combination of increased fission/absorption and increased number of neutrons/fission yields <u>more excess neutrons</u> from Pu-239
  - Enables “breeding” of fissile material
- In a fast spectrum, U-238 capture is more prominent
  - <u>Higher enrichment</u> (TRU/HM) is required (next viewgraph)
  - Enhances internal conversion
- Reduced parasitic capture and improved neutron balance
  - Allows the use of conventional stainless steel structures
  - Slow loss of reactivity with burnup
    - *Less fission product capture and more internal conversion*
- The lower absorption cross section of all materials leads to a much longer neutron diffusion length (10-20 cm, as compared to 2 cm in LWR)
  - <u>Neutron leakage is increased</u> (>20% in typical designs)
  - Reflector effects are more important
  - Heterogeneity effects are relatively unimportant

Impact of Energy Spectrum on Enrichment and Depletion Behavior

Reaction	Thermal Concepts			Fast Concepts		
	PWR	VHTR	SCWR	SFR	LFR	GFR
U238c	0.91	**4.80**	0.95	0.20	0.26	0.32
Pu239f	89.2	164.5	138.8	1.65	1.69	1.90
P239f/U238c	97.7	34.3	146.6	**8.14**	**6.59**	**6.00**
Fe	0.4			**0.007**		
Fission Prod.	90			**0.2**		

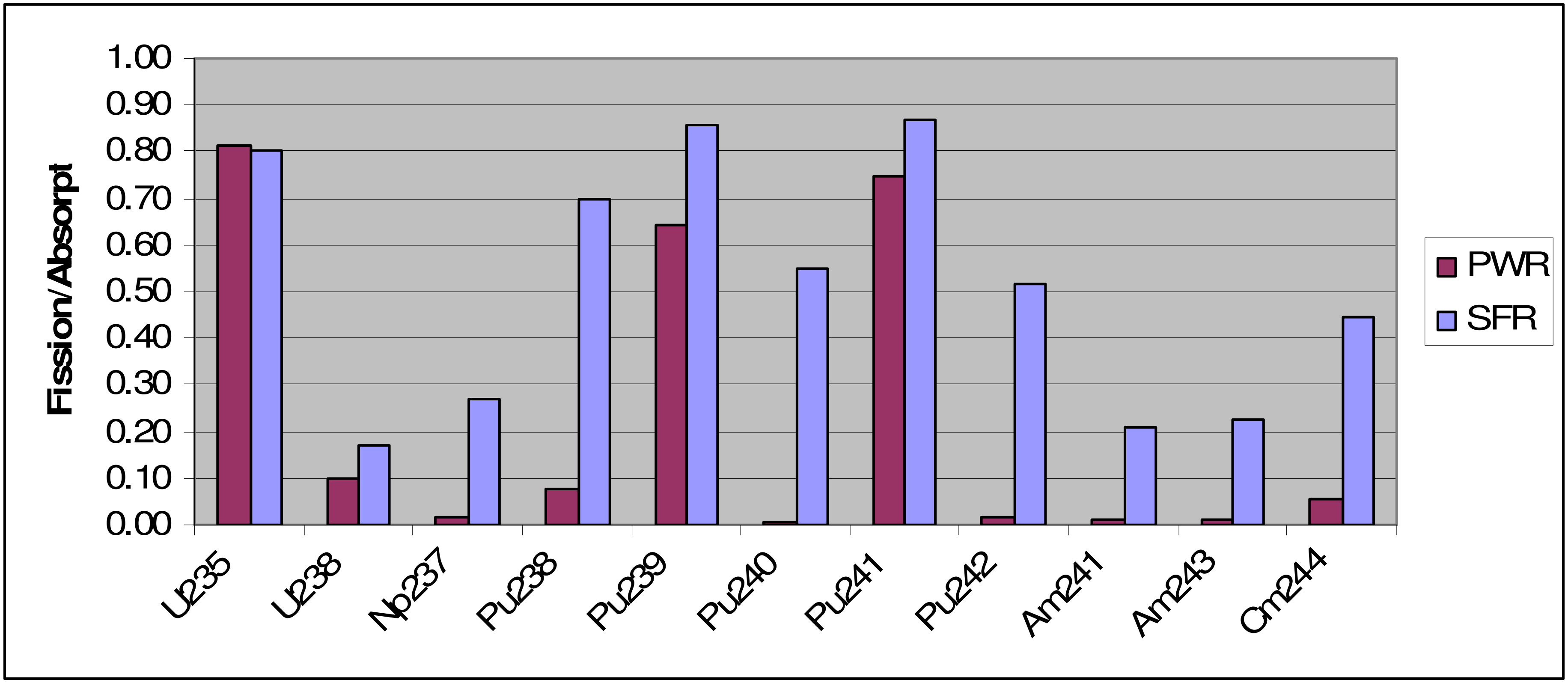
- Generation-IV fast systems have similar characteristics
- One-group XS are significantly reduced in fast system
- However, U-238 capture is much more prominent (low P239f/U238c)
  - *A much higher enrichment is required to achieve criticality*
- The parasitic capture cross section of fission products and conventional structures is much higher in a thermal spectrum (next viewgraph)

Neutron Balance

		PWR	SFR	
			CR=1.0	CR=0.5
U-235 or TRU enrichment, %		**4.2**	**13.9**	**33.3**
Source	fission	100.0%	99.8%	99.9%
	(n,2n)		0.2%	0.1%
Loss	leakage	3.5%	**22.9%**	**28.7%**
	radial	3.0%	12.3%	16.6%
	axial	0.4%	10.6%	12.1%
	absorption	96.5%	77.1%	71.3%
	fuel	76.7%	71.8%	62.2%
	(U-238 capture)	(27.2%)	(31.6%)	(17.1%)
	coolant	**3.4%**	0.1%	0.1%
	structure	0.6%	3.7%	3.7%
	fission product	**6.8%**	1.5%	2.4%
	control	9.0%	0.0%	2.9%

- Conversion ratio defined as ratio of TRU production/TRU destruction
  - Slightly different than traditional breeding ration with fissile focus

Impact of Energy Spectrum on Fuel Cycle (Transmutation) Performance



- Fissile isotopes are likely to fission in both thermal/fast spectrum
  - Fission fraction is higher in fast spectrum
- Significant (up to 50%) fission of fertile isotopes in fast spectrum

Net result is more excess neutrons and less higher actinide generation in FR

Spectral Comparison of Isotopic D-factors

Isotope	Thermal Concepts			Fast Concepts		
	PWR	VHTR	SCWR	SFR	LFR	GFR
U-235	0.65	0.53	0.70	1.04	0.92	0.84
U-238	0.02	-0.26	-0.01	0.89	0.71	0.62
Np-237	-0.96	-1.11	-1.03	0.88	0.65	0.51
Pu-239	0.83	0.72	0.80	1.71	1.59	1.45
Pu-240	-0.04	-0.12	-0.09	1.28	1.04	0.94
Pu-241	0.95	0.88	0.90	1.42	1.29	1.27

- D-factor measures the neutron balance to completely fission a given isotope
  - Positive value indicates excess neutrons are generated
- Fast systems have favorable neutron balance for all TRU isotopes
  - Thermal reactor only for the fissile isotopes
- Thus, fast systems can efficiently convert U-238 and consume the actinides, while a fissile source is require to sustain the thermal conversion

Equilibrium Composition in Fast and Thermal Spectra

Isotope	Once-Through	Fast U-238	Thermal U-238
Np237	0.048	0.008	0.002
Pu238	0.024	0.014	0.046
Pu239	0.476	0.666	0.388
Pu240	0.225	0.243	0.197
Pu241	0.106	0.021	0.111
Pu242	0.066	0.018	0.085
Am241	0.034	0.021	0.019
Am242m	0.000	0.001	0.001
Am243	0.015	0.005	0.033
Cm242	0.000	0.000	0.002
Cm244	0.005	0.002	0.055
Cm245	0.000	0.000	0.018
Cm246	0	0.000	0.031
Cm247	0	0.000	0.004
Cm248	0	0.000	0.006

- Equilibrium higher actinide content much lower in fast spectrum system
- Generation of Pu-241 (key waste decay chain) is suppressed
- However, if starting from once-through LWR composition (e.g., burner reactor) the higher actinide content will be higher than the U-238 equilibrium

Fuel Cycle Implications

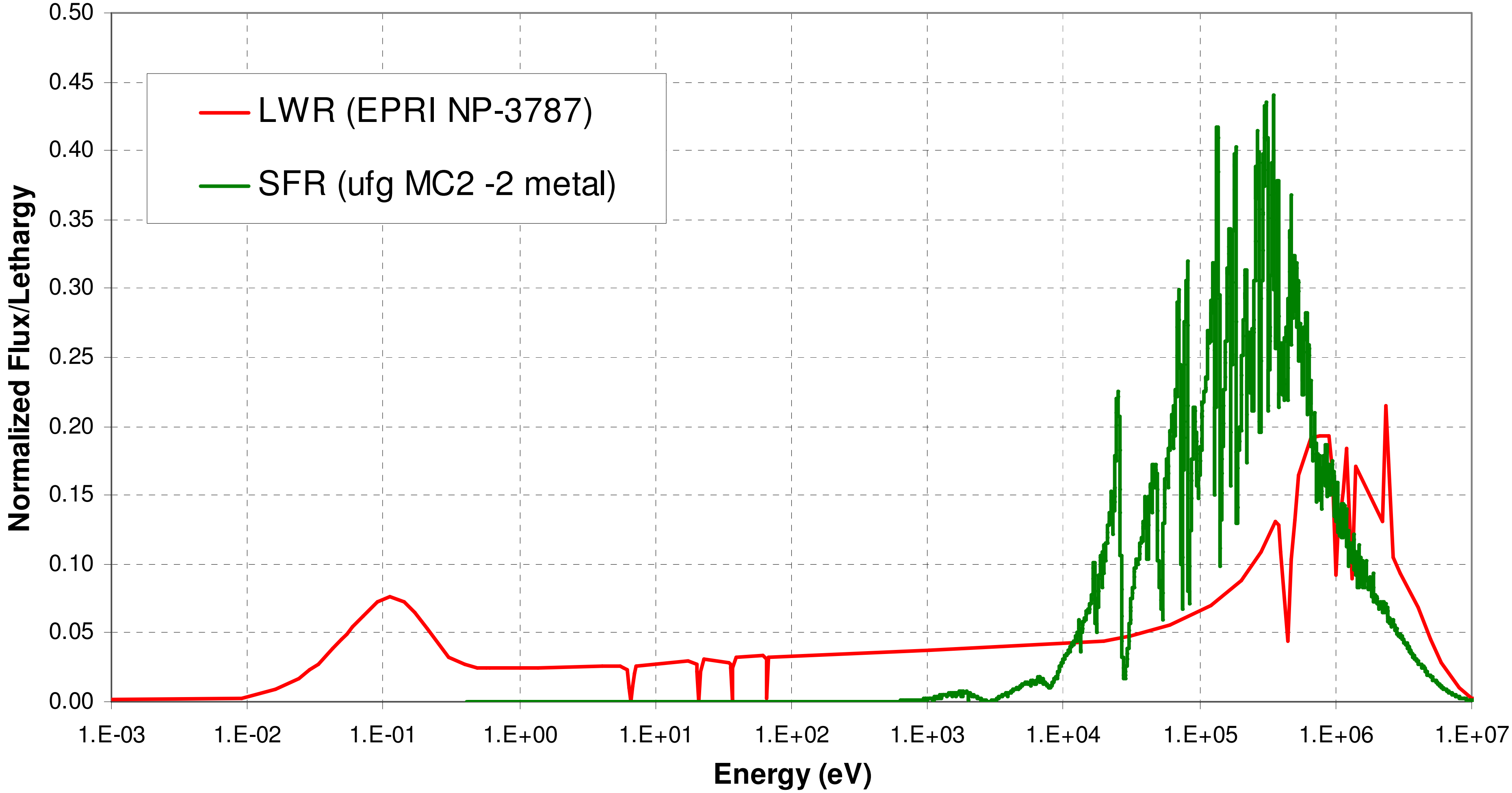
The physics distinctions facilitate different fuel cycle strategies

- **Thermal reactors** are typically configured for once-through (open) fuel cycle
  - They can operate on low enriched uranium (LEU)
  - They require an external fissile feed (neutron balance)
  - Higher actinides must be managed to allow recycle
    - *Separation of higher elements – still a disposal issue*
    - *Extended cooling time for curium decay*
- **Fast reactors** are typically intended for closed fuel cycle with uranium conversion and resource extension
  - Higher actinide generation is suppressed
  - Neutron balance is favorable for recycled TRU
    - *No external fissile material is required*
    - *Can enhance U-238 conversion for traditional breeding*
    - *Can limit U-238 conversion for burning*

Computational Methods

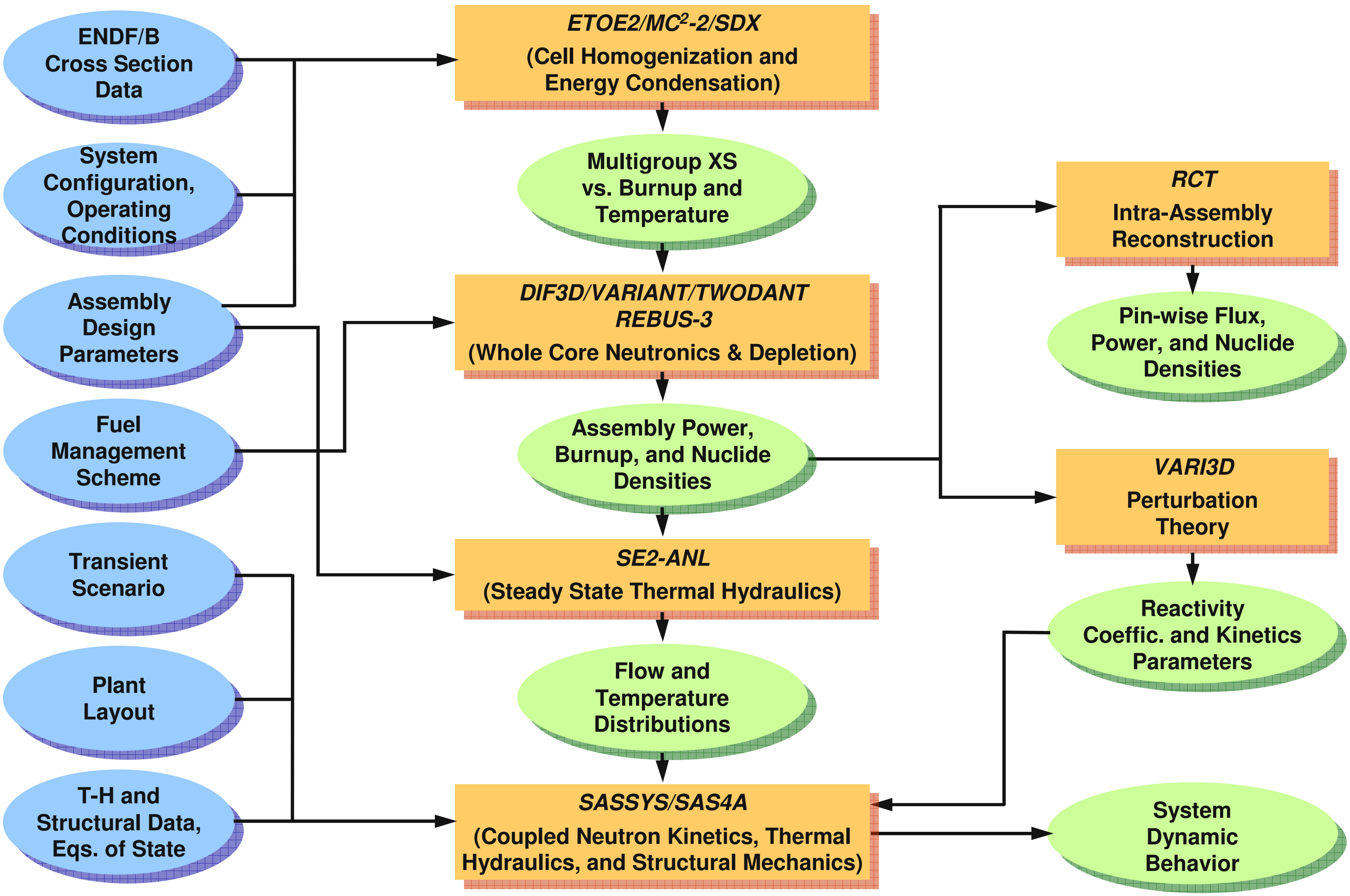
- Many of the assumptions employed in traditional LWR methods do not apply
  - Lack of a 1/E type spectrum as a basis for the calculation of resonance absorption
    - $E\varphi(E)$ *strongly decreases with decreasing energy in FR*
  - Up-scattering resulting from the thermal motion of the scattering nuclei may be neglected
  - Inelastic, (n,2n), anisotropic scatterings are of great importance
  - Long mean free path implies global coupling
    - *Local reactivity effects impact entire core*
  - The energy range where neutrons induce fission and the energy range where the fission neutrons appear strongly overlap
- Other physics consideration have high priority in FR methods
  - Detailed energy modeling for resonance structure (core/reflector)
  - Transport and anisotropy effects more important at high energy
- In general, a distinct set of physics analysis and core design tools with tailored assumptions was developed for fast reactor analysis

Comparison of LWR and SFR Spectra



- In LWR, most fissions occur in the 0.1 eV thermal "peak"
- In SFR, moderation is avoided – no thermal neutrons

Existing Sodium-Cooled Fast Reactor Code Suite



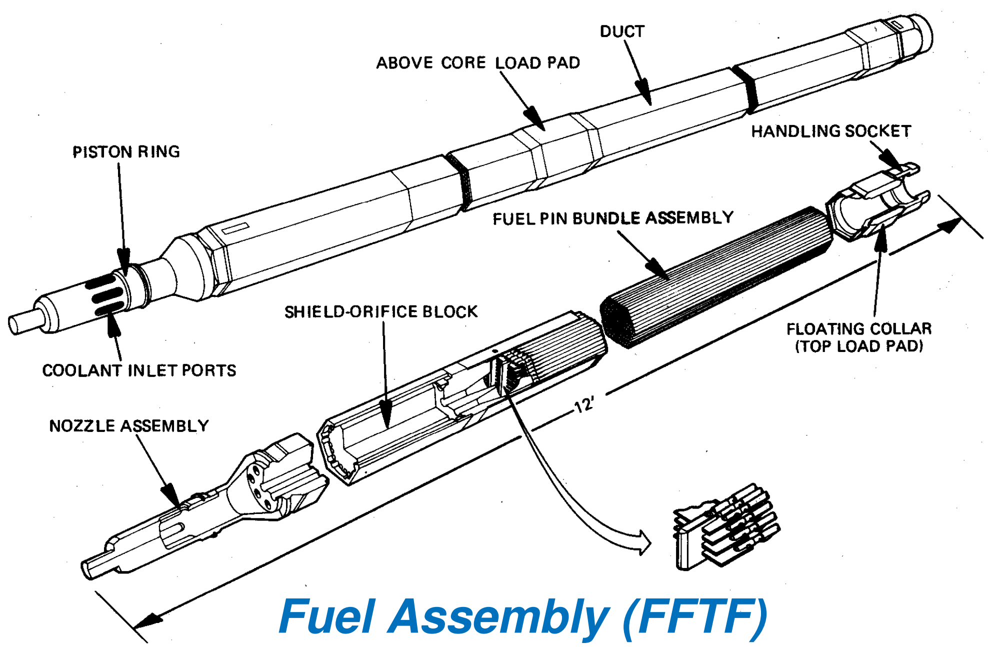
Typical Neutronics Analysis Techniques

- Design calculations typically performed using ~33 energy groups
  - Cross sections self-shielded at ultrafine group (~2000+) level
  - Spatial collapse to form regional broad group cross sections
- Design calculations utilize nodal diffusion method
  - Nodal transport techniques also available
  - Continuous energy Monte Carlo for benchmarking
  - Advanced transport methods (generalized geometry and many energy groups) being developed in GNEP advanced simulation campaign
- Depletion calculations include extensive external cycle modeling
  - Several axial depletion regions within assembly (pin-wise for processing)
  - Conventional FR fuel management does not shuffle fuel
- Current tools are adequate to begin the design process
  - Extensive critical experiment and reactor operation database exists
  - Validation and capabilities have evolved in parallel
  - Formal and comprehensive documentation needs to be done
  - This activity should support evaluation of advanced methods

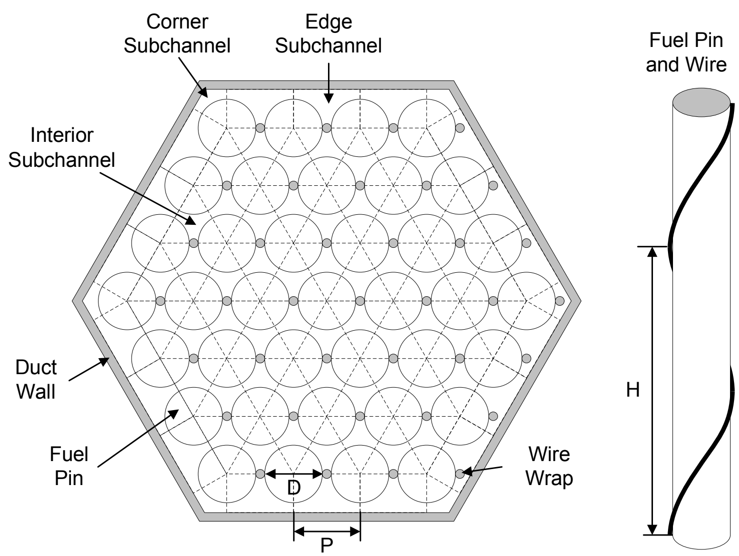
Typical Design Specifications of LWR and SFR

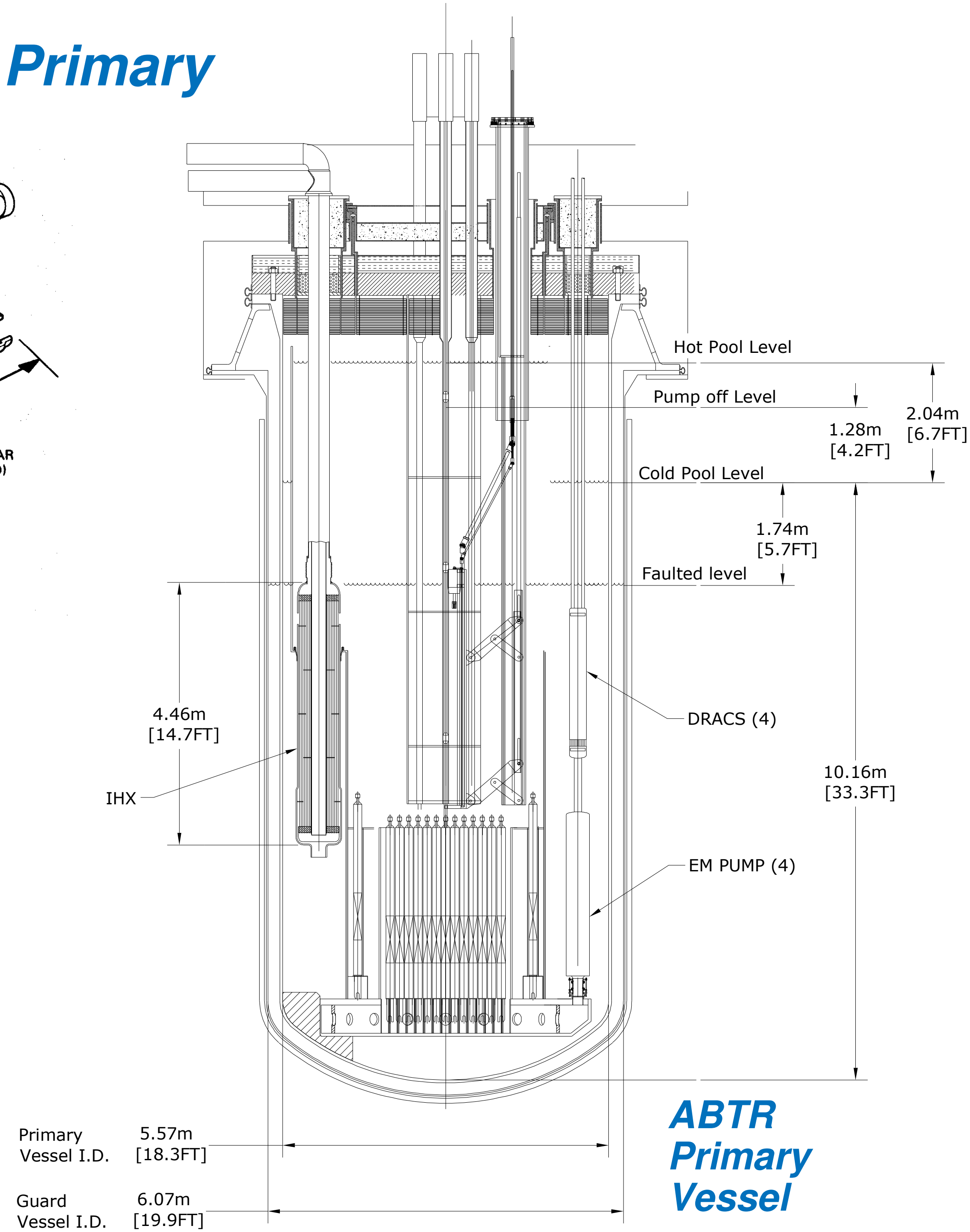
			PWR	SFR
General	Specific power (kWt/kgHM)		786 (U-235)	556 (Pu fissile)
	Power density (MWt/m^3)		102	300
Fuel	Rod outer diameter (mm)		9.5	7.9
	Clad thickness (mm)		0.57	0.36
	Rod pitch-to-diameter ratio		1.33	1.15
	Enrichment (%)		~4.0	~20 Pu/(Pu+U)
	Average burnup (MWd/kg)		40	100
Thermal Hydraulic	Coolant	pressure (MPa)	15.5	0.1
		inlet temp. (℃)	293	332
		outlet temp. (℃)	329	499
		reactor Δp (MPa)	0.345	0.827
	Rod surface heat flux	average (MW/m^2)	0.584	1.1
		maximum (MW/m^2)	1.46	1.8
	Average linear heat rate (kW/m)		17.5	27.1
	Steam	pressure (MPa)	7.58	15.2
		temperature (℃)	296	455
Control	Control Rods		0.07 (10.8 $)	Primary/Secondary (10$/3$)
	Chemical Shim		0.25 (38.5 $)	

Sodium-Cooled Fast Reactor Primary Vessel and Fuel Assembly



Fuel Assembly (FFTF)





ABTR Primary Vessel

Typical Core Design Volume Fractions

Material	PWR	SFR		LFR	GFR
		CR=1.0	CR=0.5		
Fuel	30	40	30	34	30
Coolant	59	35	44	55	57
Structure	11	25	26	11	13

- For conventional fast reactor, fuel volume fraction (VF) maximized
  - Tightly packed pin lattice
  - High volume fraction blankets to introduce additional U-238
- For burner design, TRU production reduced by lower fuel VF
  - Smaller pins to yield increased coolant VF
- For LFR, higher coolant VF is required to reduce coolant velocity
  - Needed for oxygen control to prevent cladding erosion
- For GFR, higher coolant VF is required because of inferior heat transfer
  - Trade-off between pumping power and neutronic performance

Early Fast Reactors and Fuel Forms

Original choice was high density metal fuel (for breeding)

- First usable nuclear electricity– **EBR-I in 1951**
- EBR-II (1963), Fermi (1963), DFR (UK, 1959) all used metal fuel
- Early designs experienced severely limited fuel burnup because of fuel swelling (U-10Mo burnup of 3 GWd/MT for Fermi)

U.S. and international programs switched to oxide fuel in the late 1960s

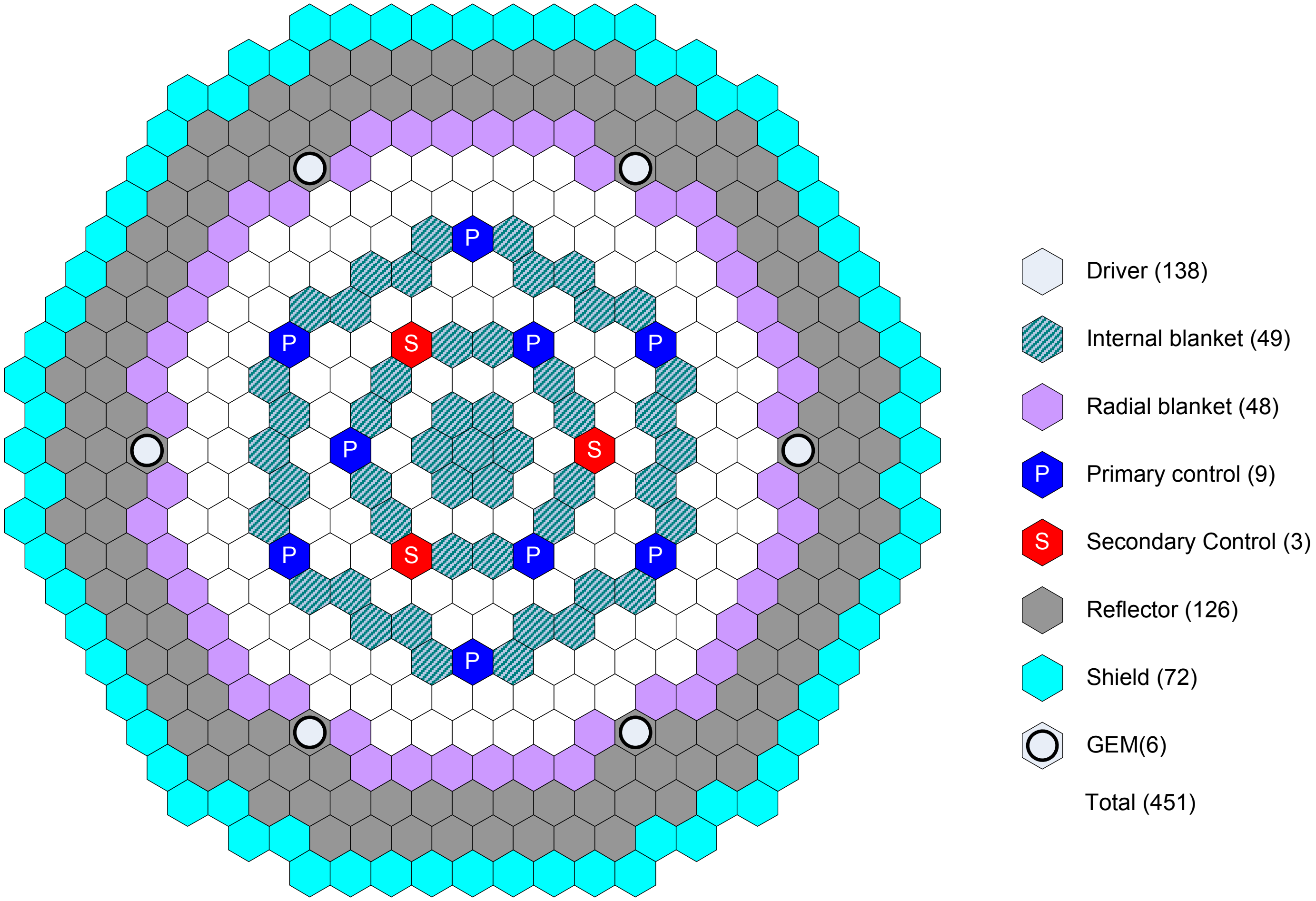
- Low swelling and successful Navy oxide fuel experience → high burnup
- Fast Flux Test Facility (400 MWt) operated with oxide from 1980 to 1992

- EBR-II (20 MWe) continued metal fuel development from 1963 to 1994
  - *Solved burnup limitation by allowing adequate space for fuel swelling*
  - *Demonstrated peak burnup comparable to oxide fuel (200 GWd/MT)*

Fast Reactor Fuel Options

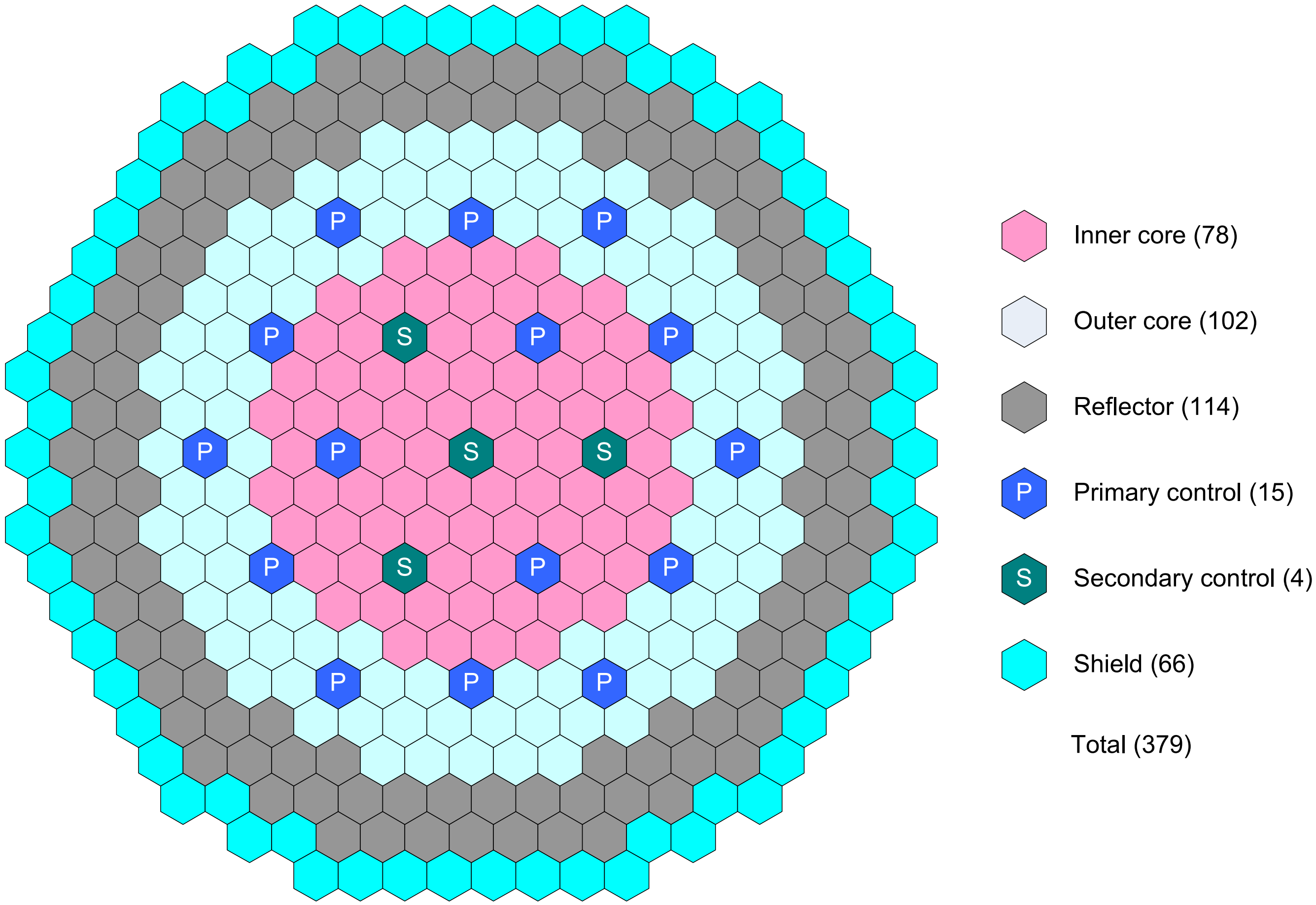
Fast Reactor Fuel Type Fresh Fuel Properties	Metal U-20Pu-10Zr	Oxide UO_2-20PuO_2	Nitride UN-20PuN	Carbide UC-20PuC
Heavy Metal Density, g/cm^3	14.1	**9.3**	13.1	12.4
Melting Temperature, °K	**1350**	3000	3035*	2575
Thermal Conductivity, W/cm-°K	0.16	**0.023**	0.26	0.20
Operating Centerline Temperature at 40 kW/m, °K, and (T/T_{melt})	1060 (0.8)	2360 (0.8)	1000 **(0.3)**	1030 **(0.4)**
Fuel-Cladding Solidus, °K	**1000**	1675	1400	1390
Thermal Expansion, 1/°K	17E-6	12E-6	10E-6	12E-6
Heat Capacity, J/g°K	0.17	0.34	0.26	0.26
Reactor Experience, Country	US, UK	RUS, FR, JAP US, UK		IND
Research & Testing, Country	US, JAP, ROK, CHI	RUS, FR, JAP, US, CHI	US, RUS, JAP	IND

Conventional 1000 MWt SuperPRISM (Metal Core)



- Internal and external blankets allocated
  - Result in conversion ratio of ~1
- Only 12 control rod locations with very low burnup reactivity losses
- Blanket, two row reflector, and boron carbide for radial shielding

Burner 1000 MWt Preliminary ABR Burner Design



- Two enrichment zones to reduce radial power peaking
- No blankets allocated for conversion ratio < 1
- Additional (20) control rod locations for burnup reactivity losses
- Similar radial shield configuration

Core Performance and Mass Flow

		SPRISM – Metal		1000 MWt ABR	840 MWt LCFR	3500 MWt US-Eur
		Breakeven	Breeder			
Cycle length, month		23	23	12	6	12
Number of batches		4	4	4	6	3
Average TRU enrichment, %		~11	~9	21.9	~50	14
Fissile/TRU conversion ratio		1.05 / -	1.22 / -	0.84 / 0.73	- / 0.25	1.13 / -
HM/TRU inventory at BOC, MT		26.1 / 3.1	36.3 / 3.2	13.2 / 2.9	4.6 / 2.25	59.5 / 5.9
Average/peak burnup, MWd/kg		106 /149	103/145	93 / 138	177 / 321	80 / 114
Peak Fast Fluence, 10^{23} n/cm^2		3.9	3.9	4.0	4.0	3.1
Burnup reactivity swing (%Δk)		0.12	-0.31	3.5	4.3	-0.62
Average linear power, kW/m		18.9	18.3	23.3		
Peak linear power, kW/m		30.4	29.8	37.2	45.4	45.4
Mass flow (kg/year)	Uranium			-255.8	-75	-1216
	Fissile Pu	19.3	69.9	-44.0		124
	MA			-6.5		
	TRU	33.6	84.6	-81.9	-193	158

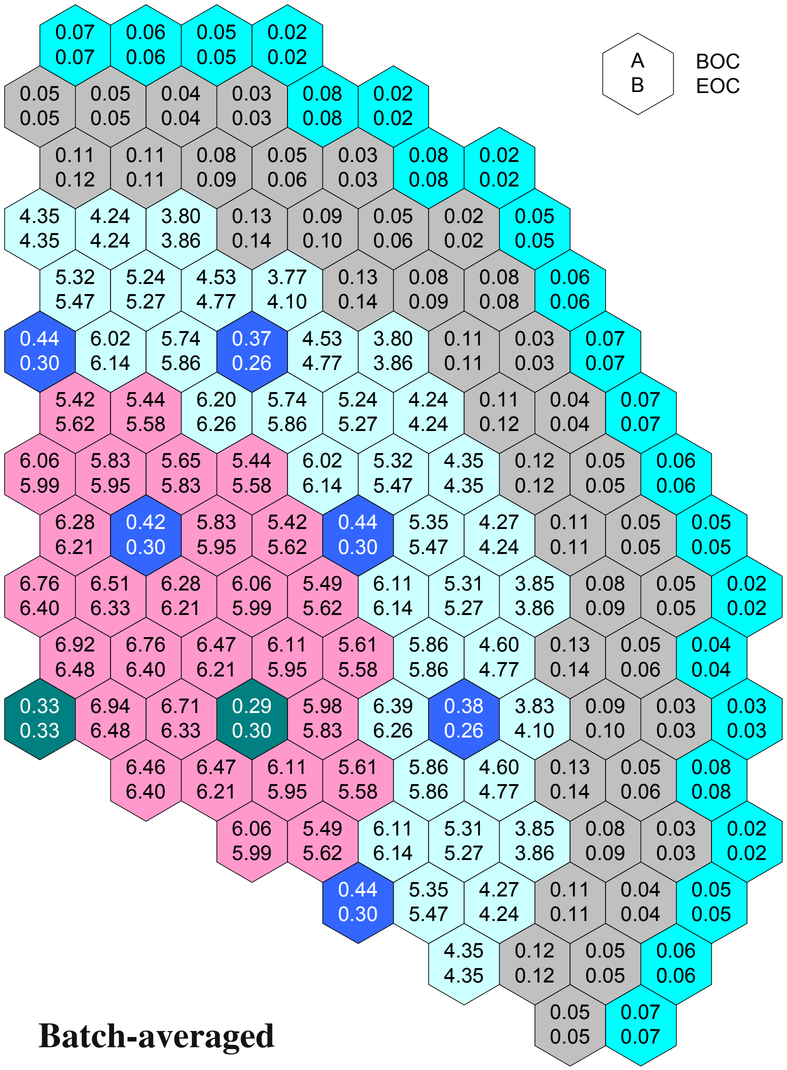
Burner Design Challenges

- Radial blanket typically replaced by reflector
  - Many criticals (BFS-62, MUSE-4) exhibit problems in accurate prediction of reaction rates in the immediate reflector region
  - Spectral and directional transitions are hard to model
  - Important for shielding and bowing (safety) considerations
- High leakage configurations also challenge design methods
  - Transport effects are magnified
  - Key reactivity coefficients (e.g., void worth) impacted
- Current GNEP strategy keeps the grouped TRUs together
  - Minor actinides present in fresh fuel
  - If low conversion ratio, the TRU enrichment could be high
  - However, plutonium remains the dominant fission source (fission contributions on next viewgraph)
  - For dedicated minor actinide burners (or targets), uncertainty of the basic MA cross section data becomes important

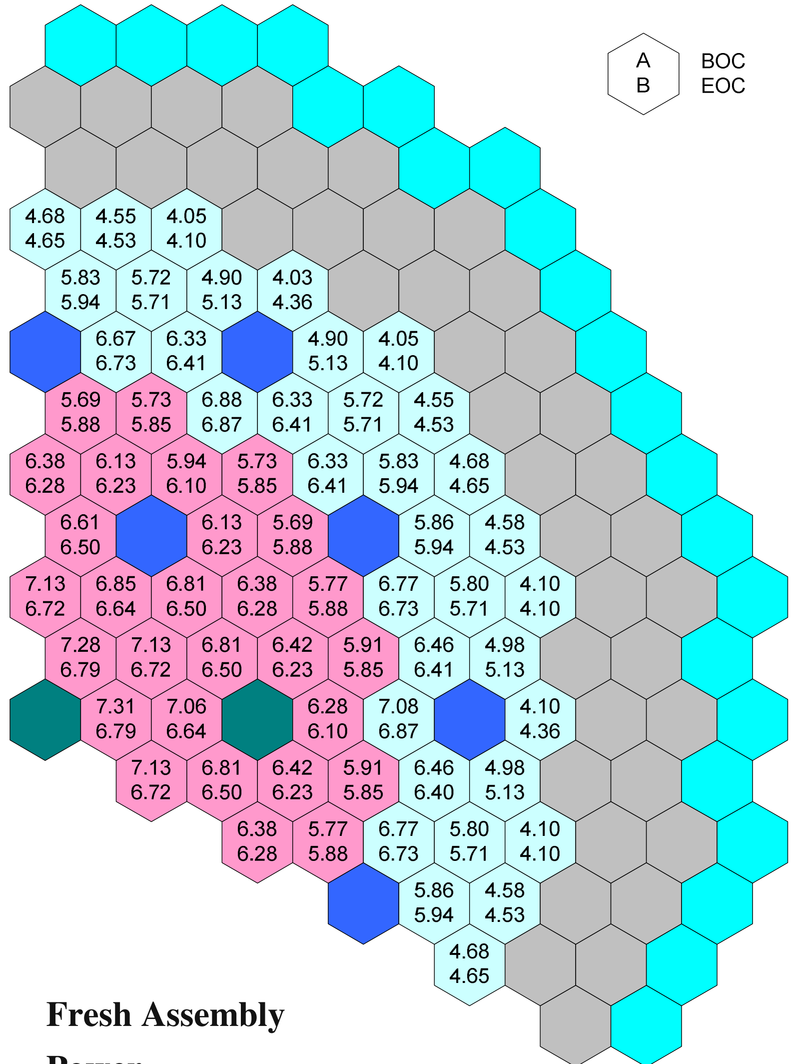
Isotopic Fission Fractions

	PWR	SFR	
		CR=1.0	CR=0.5
U-235 or TRU enrichment, %	**4.2**	**13.9**	**33.3**
U-235	59.8%	0.2%	0.2%
U-238	6.2%	15.7%	8.3%
Np-237		0.2%	0.6%
Pu-238		0.8%	3.4%
Pu-239	29.8%	70.7%	57.0%
Pu-240		6.7%	11.4%
Pu-241	4.0%	4.5%	11.9%
Pu-242		0.4%	2.5%
Am241		0.3%	0.8%
Am-242m		0.2%	0.6%
Am-243		0.1%	0.7%
Cm-244		0.1%	0.9%
Cm-245		0.1%	1.2%

Assembly Power (MW) of 1000 MWt ABR – Startup Core



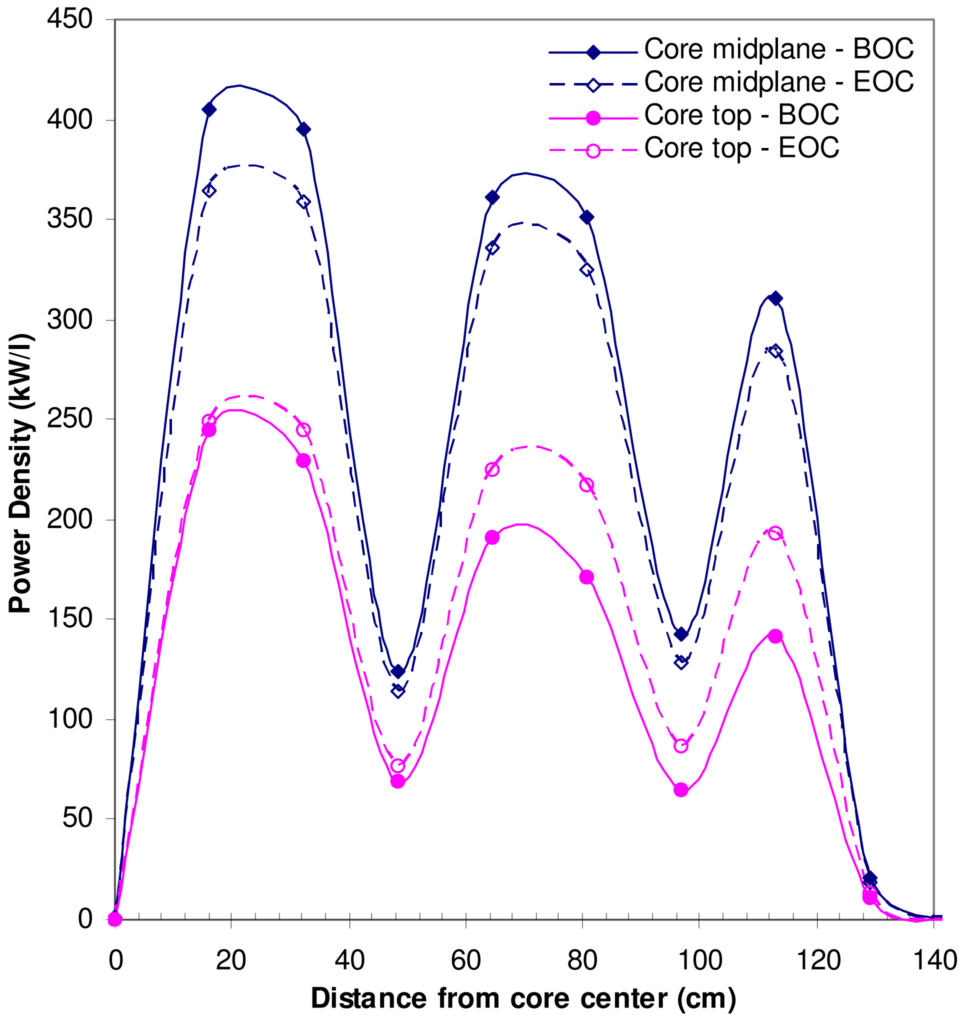
Batch-averaged Assembly Power

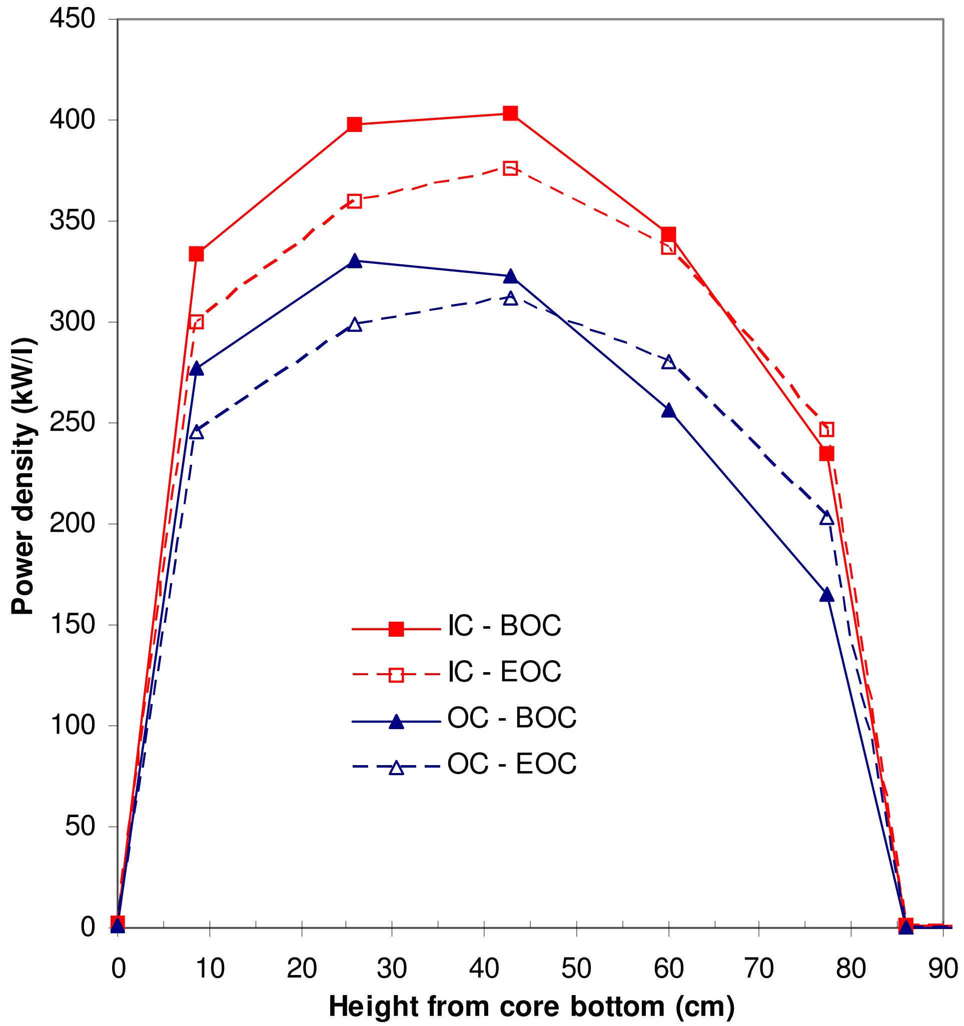


Fresh Assembly Power

- Power peaking factor (BOC/EOC) = 1.43 /1.39

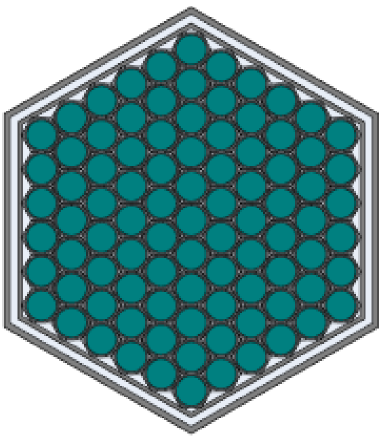
Power Profiles of 1000 MWt ABR – Startup Core

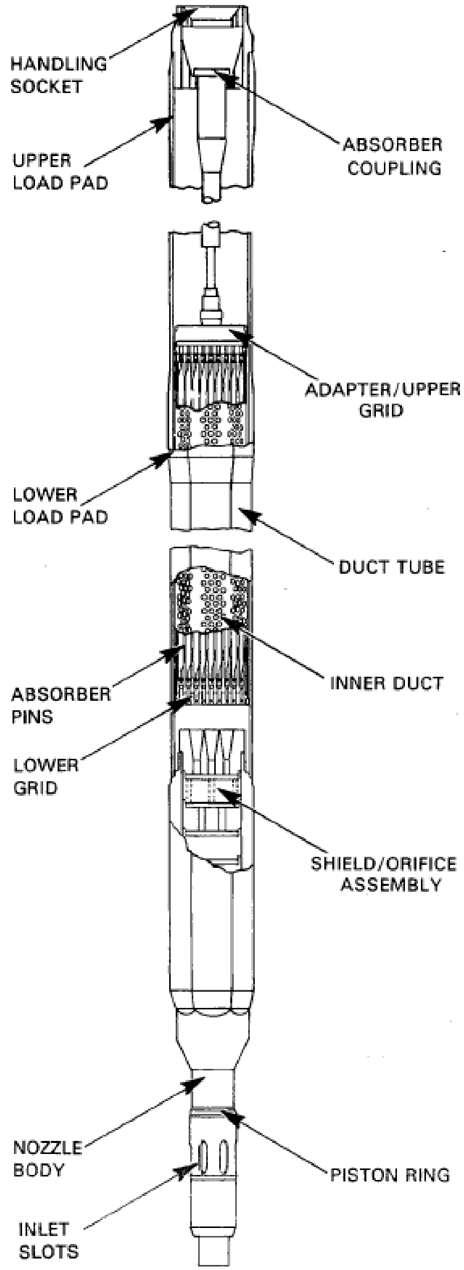


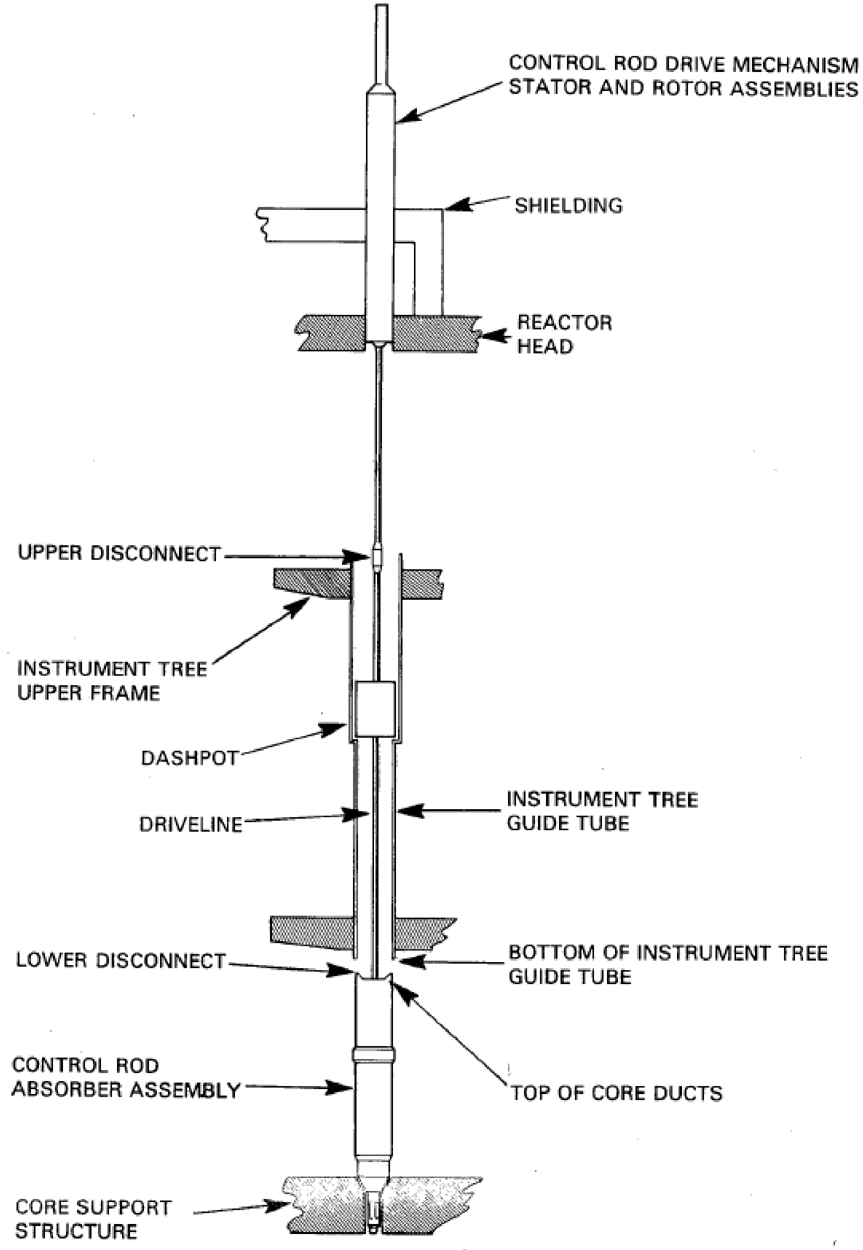


- ❑ **Bottom skewed axial distribution** (control assembly tip position at BOC = 57 cm)
- ❑ **Axial power peaking factor at BOC/EOC = 1.20/1.16**

Control Assembly







Control System Requirements

- Typical design utilizes ducted bundle of absorber pins within the coolant duct
- Sufficient reactivity worth to bring reactor
  - From any operating condition
    - *Overpower condition*
    - *Reactivity fault*
  - To cold sub-critical at refueling temperature
  - With most reactive control assembly stuck at full power operating condition
- Hold down excess reactivity for fuel cycle
  - Fuel burnup
  - Axial growth of metal fuel
- Accommodate reactivity uncertainties
  - Criticality and fissile loading
  - Temperature defect, burnup reactivity, and fuel growth

Control Requirement and Shutdown Margin ($) of Primary System

	1000 MWe LMFBR [a)]	975 MWt CRBRP [b)]	840 MWt ALMR/95	1000 MWt ABR
Number of control assemblies	13	15	10	15
Total worth	20.0	31.8	20.0	39.5
Worth of 1 stuck rod	3.0	2.8	2.3	4.3
Reactivity worth available	17.0	29.0	17.7	35.2
Control requirement	12.0	26.5	12.7	17.9
- Temperature defect	2.4	3.2	1.1	1.4
- Burnup reactivity loss	5.0	18.2	7.2	10.4
- Fuel axial growth			0.5	1.2
- Overpower margin	0.3	0.2	0.1	0.1
- Reactivity fault	3.0	2.8		0.7
- Uncertainties	1.3	2.1	2.0	2.3
- Other margin [c)]			1.8	1.8
Shutdown margin	5.0	2.5	5.0	17.3

a) GEFR-00392 (1978)
b) Early homogeneous design (1974)
c) ALMR/95 requires additional margins such as ATWS reactivity and fast runback margin

Requirements of Secondary System

- Sufficient reactivity worth to shutdown reactor
  - From any operating condition
    - *Including reactivity fault of one primary control assembly*
  - To hot standby condition
  - With most reactive control assembly stuck at full power operating condition
- Not necessary to duplicate primary system capability
  - Hold down excess reactivity for fuel cycle
  - Reactivity uncertainties

Control Requirement and Shutdown Margin ($) of Secondary System

	1000 MWe LMFBR [a)]	975 MWt CRBRP [b)]	840 MWt ALMR/95	1000 MWt ABR
Number of control assemblies	6	4	3	4
Total worth	8.0	8.4	9.4	9.5
Worth of 1 stuck rod	2.5	2.0	3.7	2.6
Reactivity worth available	5.5	6.4	5.7	6.9
Control requirement	5.0	6.3	1.7	1.9
- Temperature defect	1.6	2.4	1.1	1.4
- Overpower margin	0.3	0.2	0.3	0.2
- Temp defect uncertainty		0.9	0.3	0.3
- Reactivity fault or others	3.1	2.8		
Shutdown margin	0.5	0.1	4.0	5.0

a) GEFR-00392 (1978)
b) Early homogeneous design (1974)

Reactivity Feedback Coefficients

- The reactivity coefficients further define the physics of system
  - Kinetics parameters
  - Response to a variety of perturbations
- Feedback coefficients are computed for a specific design (geometric and material) configuration
  - Typically evaluated for BOEC and EOEC composition
- Typical set of whole-core reactivity coefficients
  - Computed with perturbation theory for spatial distributions
    - *Delayed neutron fraction and prompt lifetime*
    - *Sodium density coefficient and void worth*
    - *Fuel and structural Doppler coefficient*
    - *Fuel and structural worth distributions*
  - Computed by eigenvalue difference ($\Delta k/k$)
    - *Uniform axial expansion*
    - *Uniform radial expansion*
    - *Control rod driveline expansion*

Whole-Core Reactivity Coefficients for Different Powers

	unit	250 MWt ABTR	1000 MWt ABR	3500 MWt US-Europe
Effective delayed neutron fraction		0.0033	0.00334	0.0035
Prompt neutron lifetime	Ms	0.33	0.38	0.32
Radial expansion coefficient	¢/◦C	-0.43	-0.38	-0.21
Axial expansion coefficient	¢/◦C	-0.05	-0.05	-0.07
Sodium density coefficient	¢/◦C	0.03	0.13	0.18
Doppler coefficient	¢/◦C	-0.10	-0.13	-0.13
Sodium void worth	$	1.10	4.93	7.29* (4.98)
Sodium voided Doppler coefficient	¢/◦C	-0.07	-0.09	-0.09

- Power coefficient is quite negative
  - More negative at smaller size because of radial expansion coefficient
  - Sodium density coefficient also more positive at larger size
- Physics underlying each component will be explained
  - Void worth will be addressed separately

Delayed Neutron Fraction

- Hummel and Okrent – *Reactivity Coefficients in Large Fast Power Reactors, ANS, 1970* is a good reference for underlying physics
- Delayed neutron fraction dominated by key fission isotopes
  - Low (0.2%) for Pu-239
  - High (1.5%) for U-238
  - Between 0.3-0.5% for higher plutonium isotopes
  - Particularly low (<0.2%) for minor actinides
- Net result is 0.3-0.4% for conventional compositions
  - Slightly lower for burner designs (next viewgraph)
- Much higher for U-235 enriched systems
  - Delayed neutron fraction for U-235 is ~0.67%
- Delayed neutron fraction is an indicator of sensitivity
  - At low values, response to small changes in the reactivity is magnified and power can change more quickly
  - Feedback effects can be favorable or not depending on the transient

Low Conversion Ratio Fast Reactor Analyses

Fast reactors with closed fuel cycle can effectively manage TRU

- **Can be configured as modest breeders (CR≥1) to moderate burners (CR≥0.5) with conventional technology**
- **Low conversion ratio designs (CR<0.5) have been investigated for transmutation applications in AFCI**
  - **High enrichment fuels are required (~50% TRU/HM for CR=0.25)**
  - **Non-uranium fuel would be needed to achieve CR=0**

Conversion Ratio	TRU/HM Enrichment	Equilibrium Fraction of FR	Delayed Neutron Fraction
1.1	11%	100%	0.0034
0.75	20%	51%	0.0033
0.5	30%	34%	0.0031
0.25	50%	26%	0.0028
0.0	100%	21%	0.0020

Geometric Expansion Coefficients

Whole-core coefficients are computed by eigenvalue difference for a small change in each dimension

- Radial expansion – uniform expansion of grid plate by 1%
  - Reduction of fuel/structure densities by 1%
  - This allows more axial leakage in particular
- Axial expansion – uniform expansion of fuel by 1%
  - Reduction of fuel density by 1%
  - Allows more radial leakage
  - Also, effectively inserts the control rods which remain stationary
  - In some cases, fuel assumed bound to clad for axial expansion
- These feedbacks are very important for fast reactor transient behavior
  - Tied to different material temperatures (grid plate, fuel)
  - Thus, timing will be different

Neutron Balances of Radial and Axial Expansions

	Base Case	Radial Expansion		Axial Expansion	
	balance	balance	$\Delta\rho$ (%)	balance	$\Delta\rho$ (%)
Fission source	100.00	100.00		100.00	
(n,2n) source	0.18	0.18		0.18	
Absorption	68.89	68.93	-0.04	68.93	-0.05
Leakage	31.54	32.16	-0.63	31.61	-0.07
Radial	17.49	17.72	-0.23	17.59	**-0.10**
Axial	14.05	14.45	**-0.40**	14.02	0.03
Sum			-0.67		-0.12

- **To first order, radial expansion is an axial leakage effect, and**
- **Axial expansion is a radial leakage effect**
- **Because the height is the short dimension (more axial than radial leakage), the radial expansion coefficient is more negative**
- **Absorption effect arises from control rod absorption increases**

Coolant Density Coefficient

Coolant density coefficient computed by first-order perturbation theory to evaluate small density (temperature variation) impacts

- Spectral effect
  - Reduced moderation as sodium density decreases
  - In fast regime, this is a positive reactivity effect
    - *From Pu-239 excess neutrons and threshold fission effects*
- Leakage effect
  - Sodium density decrease allows more neutron leakage
  - This is a negative reactivity effect in the peripheral regions
- Capture effect
  - Sodium density decrease results in less sodium capture
  - This is a relatively minor effect

Void worth is evaluated using exact perturbation theory to account for shift in flux distribution and change in cross sections for voided condition

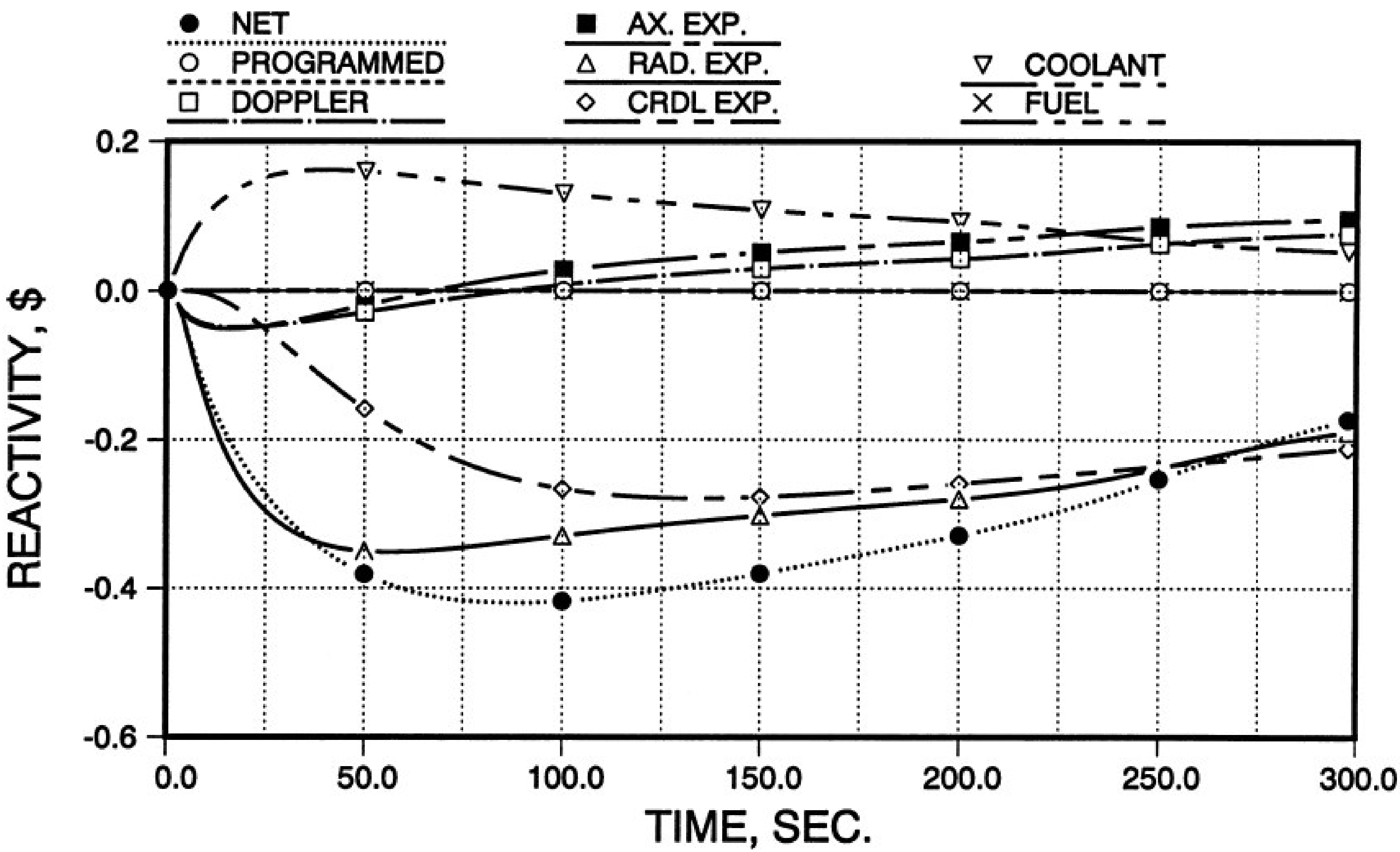
- In general, 10% more positive than the first-order density worth

Sodium Void Worth by Components ($)

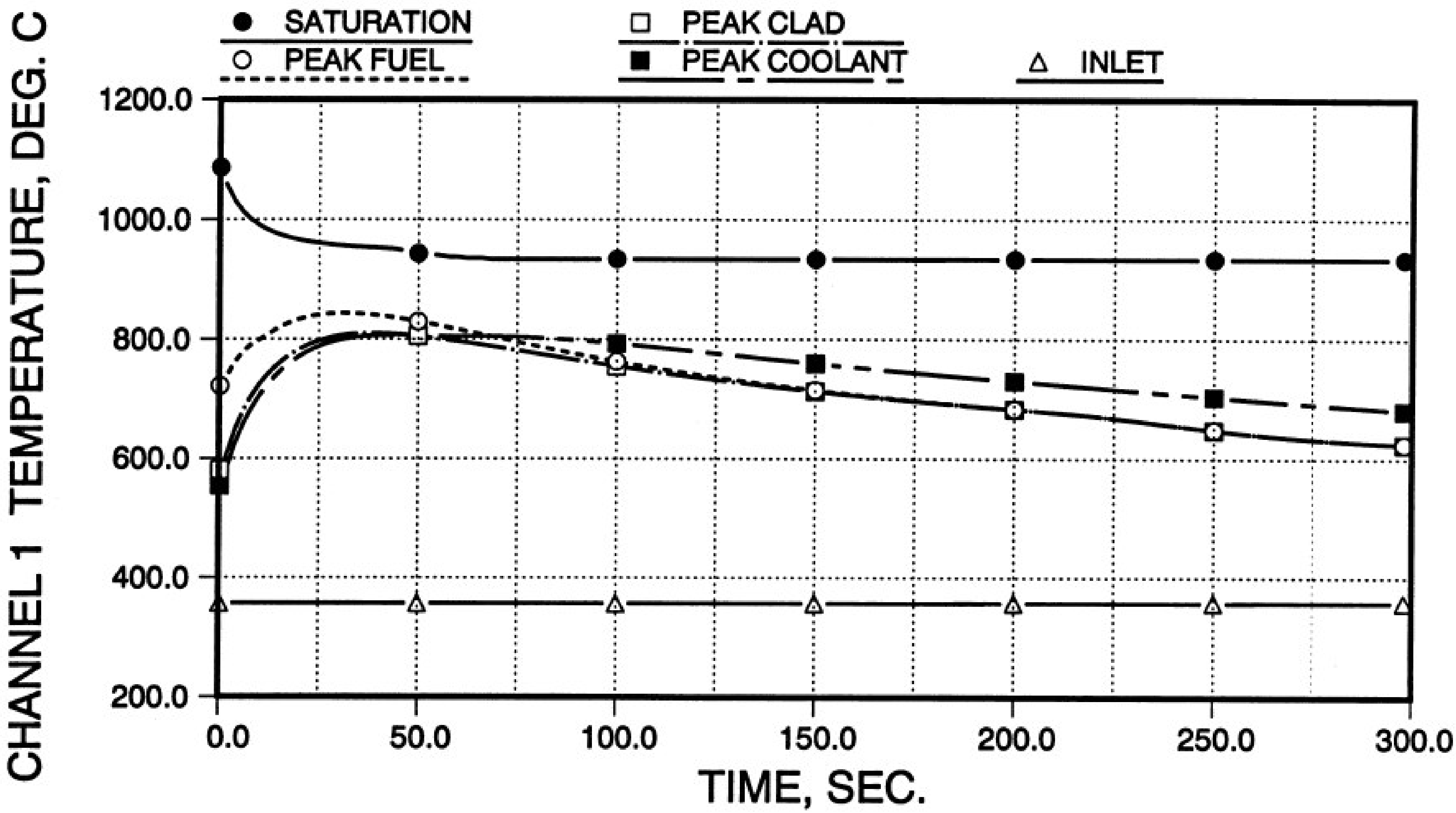
		Capture	Spectral	Leakage	Total
1000 MWt ABR (startup metal core)	BOC	0.5	9.1	-5.2	4.4
	EOC	0.5	9.9	-5.5	4.9
250 MWt ABTR (startup metal core)	BOC	0.4	6.4	-5.8	1.0
	EOC	0.4	6.6	-5.8	1.1

- Flowing sodium completely voided in ALL active and above-core regions
- Void worth tends to increase with core size
- However, difficult to conceive transient situations that reach boiling
  - Low pressure system
  - >300^{o}C margin to boiling
  - Other feedbacks are negative (see next viewgraphs)
- Extensive report on void worth reduction – Khalil and Hill, NSE, 109 (1995)

Response to a ULOF in Large 3500 MWt SFR using Metal Fuel



Response to a ULOF in Large 3500 MWt SFR using Metal Fuel



Doppler Coefficient

- Doppler coefficient arises primarily from U-238 resonance broadening
  - Enhanced by high U-238 content
    - *Reduced Doppler for high enrichment burner concepts*
  - Self-shielding effect more pronounced at low energies (keV range)
    - *Doppler enhance by spectral softening*
    - *Voided Doppler is smaller from spectral shift*
- Temperature dependence in fast spectrum is different than LWR
  - Doppler range from $1/T^{1/2}$ for large to $1/T^{3/2}$ for small resonances
  - For typical FR, an approximate 1/T dependence observed
- There is also a structural Doppler reactivity effect (~1/3 fuel Doppler)
  - However, tied to temperature of steel, not fuel (different timing)
- Doppler feedback is not helpful in all transients
  - For example, when trying to cool the fuel to shutdown condition (e.g., ULOF), it is a positive feedback
  - Conversely prompt negative feedback in UTOP trasnsient

Passive Safety Behavior

- Inherent safety trends and general criteria are explained in Wade and Fujita, *Trends versus Reactor Size of Passive Reactivity Shutdown and Control Performance, Nuclear Science and Engineering*, 103 (1989).

Brief synopsis by Cahalan at last topical seminar

- The fast reactor reactivity balance can be written as follows:

  $\delta\rho = [P(t)-1]\ \mathbf{A}\ +\ [P(t)/F(t)\ -1]\ \mathbf{B} + [\delta T_{in}(t)]\ \mathbf{C} + \delta\rho_{external}$

  where $P(t)$ = normalized reactor power

  $F(t)$ = normalized core coolant flow

  $\delta T_{in}(t)$ = change in coolant temperature at the core inlet

  $\delta\rho_{external}$ = externally applied change in reactivity (control rods, etc.)

  - the relative importance of each of these terms is determined by the grouped reactivity feedback parameters, A, B, and C

Reactivity Feedback Coefficients

- The reactivity feedback coefficients that form the three parameters A, B, and C are associated with the reactor core, and depend on fuel type, fuel volume fraction, coolant volume fraction, etc.,

  $A = \alpha_{Doppler}\,\Delta T_{FC}(t=0)$

  $B = [\alpha_{Doppler} + \alpha_{NaCoolant} + \alpha_{AxialExp.} + a_1\alpha_{RadialExp} + a_2\alpha_{ControlRod}]\Delta T_C(t=0)/2$

  $C = [\alpha_{Doppler} + \alpha_{NaCoolant} + \alpha_{AxialExp.} + b_1\alpha_{RadialExp} + b_2\alpha_{ControlRod}]$

  where $\alpha_{Doppler}$ = Doppler coefficient

  $\alpha_{NaCoolant}$ = Sodium coolant density *coefficient*

  $\alpha_{AxialExp.}$ = Fuel axial expansion coefficient

  $\alpha_{RadialExp}$ = Core radial expansion coefficient

  $\alpha_{ControlRod}$ = Control rod driveline expansion coefficient

  $\Delta T_{FC}(t=0)$ = Steady-state temperature difference, fuel to coolant

  $\Delta T_C(t=0)$ = Steady-state coolant temperature rise, inlet to outlet

  a_1, a_2, b_1, b_2 = geometric parameters

Passive Safety Criteria

Criteria established for acceptable asymptotic core outlet temperatures for possible unprotected accident scenarios

- $A/B < 1$ for passive control of pump and BOP-induced accident scenarios
- $1 < C\Delta T_c/B < 2$ for LOF, pump overspeed and chilled inlet
- $\delta\rho_{TOP}/B < 1$ for TOP performance
- Comparison of the whole-core reactivity coefficients to these criteria gives and indicated that the concept has favorable passive safety features
  - Detailed safety analyses required to confirm performance and margins

Summary and Conclusions

- Fast reactor physics are quite different from thermal reactor behavior
  - Better neutron balance (flexible actinide management)
  - Higher enrichment is required to compensate U-238 capture
  - Neutron leakage is increased
- Typical fast reactor core designs were reviewed
  - Traditional blanketed breeder, moderate burner with no blanket, low conversion ratio option (high fuel enrichment) configurations
  - Reactor performance, high power density, burnup, fluence
  - Reactivity compensation and control rod requirements
- Reactivity coefficients were discussed
  - Expansion coefficients prominent because of high leakage
  - Positive sodium density (and void) coefficient
  - Overall favorable passive performance for complete set of feedbacks